

February 2, 2005

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Duke Energy Corporation
Catawba Nuclear Station Unit 1
Docket No. 50-413
Licensee Event Report 413/2004-004 Revision 0
Spurious Moisture Separator Reheater High Level
Actuation Resulting in an Automatic Turbine Trip
and Reactor Trip

Attached please find Licensee Event Report 413/2004-004
Revision 0, entitled "Spurious Moisture Separator Reheater
High Level Actuation Resulting in an Automatic Turbine Trip
and Reactor Trip".

This Licensee Event Report does not contain any regulatory
commitments. This event is considered to be of no
significance with respect to the health and safety of the
public. Questions regarding this Licensee Event Report
should be directed to G. K. Strickland at 803-831-3585.

Sincerely,



D. M. Jamil

Attachment

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xc:

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LICENSEE EVENT REPORT (LER)

(See reverse for required number of
digits/characters for each block)

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME Catawba Nuclear Station, Unit 1	2. DOCKET NUMBER 050- 00413	3. PAGE 1 OF 6
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4. TITLE
Spurious Moisture Separator Reheater High Level Actuation Resulting in an Automatic Turbine Trip and Reactor Trip

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
12	05	2004	2004	- 004 - 00		02	02	2005	NA	
11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)										
9. OPERATING MODE 1			20.2201(b)			20.2203(a)(3)(ii)			50.73(a)(2)(ii)(B)	
			20.2201(d)			20.2203(a)(4)			50.73(a)(2)(iii)	
			20.2203(a)(1)			50.36(c)(1)(i)(A)			X 50.73(a)(2)(iv)(A)	
			20.2203(a)(2)(i)			50.36(c)(1)(ii)(A)			50.73(a)(2)(v)(A)	
			20.2203(a)(2)(ii)			50.36(c)(2)			50.73(a)(2)(v)(B)	
			20.2203(a)(2)(iii)			50.46(a)(3)(ii)			50.73(a)(2)(v)(C)	
			20.2203(a)(2)(iv)			50.73(a)(2)(i)(A)			50.73(a)(2)(v)(D)	
			20.2203(a)(2)(v)			50.73(a)(2)(i)(B)			50.73(a)(2)(vii)	
			20.2203(a)(2)(vi)			50.73			50.73(a)(2)(viii)(A)	
20.2203(a)(3)(i)			50.73(a)(2)(ii)(A)			50.73(a)(2)(viii)(B)		OTHER Specify in Abstract below or in NRC Form 366A		
10. POWER LEVEL 100%										

12. LICENSEE CONTACT FOR THIS LER

NAME G Strickland, Regulatory Compliance	TELEPHONE NUMBER (Include Area Code) (803) 831-3585
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13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTOR	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTOR	REPORTABLE TO EPIX
D1	SN	1HSL5490 and 1HSL5492	Magnetrol	Y					

14. SUPPLEMENTAL REPORT EXPECTED

YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO
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15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On December 5, 2004, at 2135 hours, with Catawba Unit 1 operating in Mode 1 at 100% power, an automatic reactor trip occurred due to a turbine trip above P-9 (69% power). The turbine trip was due to an invalid high water level indication from two of the three level switches for the 1B moisture separator reheater. The root cause of the reactor trip was due to micro-switch mechanisms on the two level switches being out of adjustment, concurrent with external vibration to the switches.

The plant response to the reactor trip remained within the limits of the Updated Final Safety Analysis Report. Major plant equipment operated as expected. This event was of no significance with respect to the health and safety of the public.

Corrective actions for this event included inspecting all Unit 1 and Unit 2 micro-switch positions, and adjusting or replacing the switch mechanisms as necessary.

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FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
Catawba Nuclear Station, Unit 1	05000413	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	2 OF 6
		2004	- 004	- 00	

NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

BACKGROUND

This event is being reported under 10 CFR 50.73 (a)(2)(iv)(A), any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature including the Reactor Protection System.

Catawba Nuclear Station (CNS) Unit 1 is a Westinghouse four-loop pressurized water reactor [EIIS: RCT]. Unit 1 was operating in Mode 1 (Power Operation) at 100% power prior to this event.

Unit 1 has four moisture separator reheaters (MSR) [EIIS: MSR]. The purpose of the MSR system is to remove the moisture and reheat the main steam from the high pressure turbine exhaust to the low pressure turbine inlet [EIIS: TRB].

Each MSR has three level switches [EIIS: LS] to detect a high level. After two-out-of-three level switches actuate for 10-seconds, the main turbine is automatically tripped. The purpose of the MSR level switches is to protect the low pressure turbine blades from water or moisture carryover. The purpose of the 10 second time delay is to prevent spurious system actuation. The level switches are calibrated on an 18 month frequency.

A turbine trip with reactor power above 69% power will initiate an automatic reactor trip.

There were no systems, structures or components out of service during this event that contributed to this event.

EVENT DESCRIPTION

(Dates and times are approximate)

12/5/04 Unit 1 reactor trip caused by turbine trip
2135 above P-9. The turbine trip was due to an
invalid high level indication in the 1B MSR.

At the time of the trip, maintenance activities were in progress in the vicinity of the MSR 1B.

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		2004	- 004	- 00	

NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

Based on interviews, these activities were not believed to be the cause of the level switch actuation.

12/5/04
2205 Plant conditions stabilized at no load conditions. The plant response to the reactor trip remained within the limits of the Updated Final Safety Analysis Report.

Major plant equipment operated as expected. The following minor issues were noted:

Steam generator 1C power operated relief valve [EIIS: PSV] lifted briefly at 1116 psig.

Steam generator 1B main feedwater control valve [EIIS: LCV] 1CF37 did not indicate fully closed. The valve automatically responded to its correct, closed position.

12/6/04
0017 Four hour notification to the NRC completed.

12/6/04 Team assembled to investigate the reactor trip.

12/7/04
0257 Unit 1 reactor critical

12/7/04
0816 Unit 1 main generator on line.

12/7/04
2124 Unit 1 at 100% power

CAUSAL FACTORS

An automatic reactor trip occurred due to a turbine trip above 69% power. The turbine trip was due to an invalid high water level indication from two of the three level switches for the 1B moisture separator reheater.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

The root cause of the reactor trip is due to micro-switch mechanisms on the two level switches being out of adjustment, concurrent with external vibration to the switches. The cause of the switch mechanisms being out of adjustment is inconclusive; however, the two probable causes are either 1) the switch mechanisms were obtained from the manufacturer out of adjustment, or 2) inadequate procedural guidance was provided to ensure the successful replacement / adjustment / testing of the switches. The second probable cause for inadequate procedural guidance is believed to be the more likely cause.

CORRECTIVE ACTIONS

Immediate:

1. Plant conditions were stabilized at no-load conditions.

Subsequent:

1. All Unit 1 and Unit 2 MSR level switches were inspected. Switches were repaired or adjusted as necessary.
2. Personnel barriers were initially placed around Unit 1 and Unit 2 MSR level switches to prevent accidental bumping of the switches. Caution signs were later placed in the vicinity of the switches and the personnel barriers were removed.
3. Valve indication for the 1B main feedwater control valve 1CF37 was repaired.

Planned:

1. Revise the applicable maintenance procedure to include appropriate guidance for switch adjustments.
2. Replace all of the current Unit 1 and Unit 2 switch mechanisms with manufacturer adjusted switch mechanisms.
3. Evaluate other, same-vendor, critical level switch applications for applicability.

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FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
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		2004	- 004	- 00	

NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

The planned corrective actions are being addressed within the Catawba Corrective Action Program. There are no NRC commitments contained in this LER.

SAFETY ANALYSIS

The solid state protection system functioned as designed. The main turbine tripped automatically in response to the MSR high level signal. Automatic reactor trip occurred as the reactor trip breakers opened within the required 150 milliseconds of receipt of "MSR Hi Level Turbine Trip" signal with power above the P-9 setpoint of 69% power. All control rods inserted normally.

Reactor coolant system pressure control functioned as expected. Pressurizer power operated relief valves and code safety valves were not challenged and did not lift during the event. The pressurizer heaters controlled primary system pressure as designed. The pressurizer spray valves did not open because the reactor coolant system pressure remained below the valve control setpoint. Pressurizer level control functioned as expected.

Reactor coolant temperature control was achieved mainly by the condenser dump valves. Steam generator 1C PORV lifted briefly at a pressure of 1116 psig, while no other steam generator PORVs or safety valves opened. The 1C PORV opening was within the actuation pressure range of 1113 to 1137 psig. The plant cooldown rate was less than the Technical Specification limit of 100 degrees F per hour.

Steam generator level control functioned as expected with makeup being provided by the motor driven auxiliary feedwater pumps. The main feedwater isolation system operated as expected with the exception of the valve indication for the 1B main feedwater control valve 1CF37. The valve fully closed in response to the feedwater isolation signal but indicated mid-position. Steam generator levels remained in the normal operating band.

In summary, the transient remained within the bounds of the Updated Final Safety Analysis Report and the post trip transient response was as expected.

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The reactor trip event has been evaluated quantitatively for risk significance considering the following:

- A reactor trip initiating event
- No additional Probabilistic Risk Assessment modeled equipment out of service

The conditional core damage probability (CCDP) for this event is calculated to be approximately $2.5E-07$. The conditional large early release probability (CLERP) is calculated to be about $3.8E-09$. These values are less than the accident sequence precursor thresholds of $1.0E-06$ and $1.0E-07$, respectively.

The dominant base case large early release frequency (LERF) sequences for Catawba involve steam generator tube rupture, interfacing systems LOCA, and seismic-initiated sequences. The reliability of the important containment safeguards systems (containment spray and hydrogen mitigation) was not impacted by the reactor trip.

This event was of no significance with respect to the health and safety of the public.

ADDITIONAL INFORMATION

Within the last three years, five other reactor trip events occurred from power operation at Catawba Unit 1 and Unit 2. None of these previous events involved spurious level switch actuation or improper level switch adjustments. Therefore, this event was determined to be non-recurring in nature.

Energy Industry Identification System (EIIS) codes are identified in the text as [EIIS: XX]. The level switch is an EPIX program reportable equipment failure.

This event does not reflect a manual reactor trip with a loss of secondary heat removal capability as monitored by the NRC performance indicator. This event did not involve a Safety System Functional Failure.

There were no releases of radioactive materials, radiation exposures, or personnel injuries associated with this event.